NON-PUBLIC?: N
ACCESSION #: 8908010458
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 1 PAGE: 1 of 7

DOCKET NUMBER: 05000413

TITLE: Manual Reactor Trip Due to Torn Gasket of Main Feedwater Valve Positioner Control Air Manifold
EVENT DATE: 06/26/89 LER #: 89-017-00 REPORT DATE: 07/26/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R. Michael Glover, Compliance TELEPHONE: 803-831-3236

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: SJ COMPONENT: HHH MANUFACTURER: M430
X SO X999 W120

REPORTABLE NPRDS: Y
N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On June 26, 1989, at approximately 0635 hours, with Unit 1 in Mode 1, Power Operation, at 100% power, 1CF28, Steam Generator (S/G) 1A Main Feedwater (CF) Control Valve, slowly began closing, causing S/G 1A level to decrease.
1. he S/G 1A level deviation alarm was received and 1CF30, S/G 1A Control Bypass Valve, was opened to supply additional CF flow. An Operator was dispatched and discovered that air was leaking from the 1CF28 control air manifold. A work request was issued to investigate and repair the air leak as S/G 1A level continued to decrease. 1CF28 eventually closed and Reactor power was reduced in an attempt to match CF flow with demand. A manual Reactor trip was initiated at 0718 hours, from 86% Reactor power, just prior to a S/G Low Low Level Reactor Trip signal. Emergency Procedure EP/1/A/5000/01, Reactor Trip or Safety Injection, was entered. The Turbine tripped due to the Reactor trip and Auxiliary Feedwater actuation occurred. A gasket in the valve's control air manifold was found to be torn, which

apparently occurred during installation of the gasket, due to inappropriate actions. The torn gasket was replaced and all other Unit 1 S/G CF control valves were inspected for air leaks.

END OF ABSTRACT

BACKGROUND

The Main Feedwater EIIS:SJ! (CF) System supplies feedwater to the four Steam Generators EIIS:HX! (S/Gs) at the temperature, pressure, and flow required to maintain proper S/G water levels commensurate with Reactor power output and Turbine EIIS:TRB! steam requirements. The CF System contains two 50% capacity variable speed Turbine Driven CF Pumps EIIS:P! (CFPTs). The CF pump speeds may be manually or automatically controlled. The CF pumps discharge through two stages of high pressure CF heaters EIIS:HTR!. Then the feedwater divides into four CF lines, each supplying one of the four S/Gs. Each of the four CF lines contains a CF Control valve EIIS:V!, a CF Control Bypass valve, two CF Check valves, and a CF Isolation valve. The CF Isolation valves function to terminate CF flow in either direction following a CF Isolation signal and also function to prevent or allow admission of feedwater to the S/Gs CF nozzles during various modes of operation. The CF Control Bypass valves normally are utilized to control CF flow up to approximately 15% load, and the CF Control valves are utilized to control CF flow from approximately 15% to 100% load. The CF Control valves

are normally automatically controlled by the S/G Level Control system to maintain proper S/G levels.

The Auxiliary Feedwater EIIS:BA! (CA) System provides an independent means of supplying feedwater to the S/Gs in addition to the CF System. The CA System functions to maintain secondary side water inventory sufficient to permit an orderly plant cooldown and to remove residual heat stored in the Reactor Coolant EIIS:AB! (NC) System for the duration of all Design Basis Events. The CA System also provides condensate grade feedwater during normal Unit startup and shutdown operations when use of the CF System at such low flow rates would be undesirable. The CA System contains two full capacity Motor EIIS:MO! Driven Pumps and one full capacity Steam Turbine Driven Pump.

Any of these pumps may be started manually or automatically. The Motor Driven

Pumps are designed to start automatically when both CFPTs trip.

EVENT DESCRIPTION

On June 26, 1989, at approximately 0635 hours, Unit 1 was in Mode 1, Power

Operation, at 100% power, when 1CF28, S/G 1A CF Control Valve, began slowly closing and causing S/G 1A level to decrease. The S/G 1A level deviation alarm was received at 0635 hours, and Control Room Operators (CROs) opened 1CF30, S/G 1A Control Bypass Valve, for additional CF flow. In addition, Reactor power was reduced to 97% by reducing Turbine load. 1CF28 was placed in manual control and CFPT 1A speed was increased. A Nuclear Equipment Operator was quickly dispatched to investigate the problem.

At approximately 0650 hours, the 1CF28 control air manifold was found to be leaking. Work Request 50912 OPS was issued to investigate and repair the leak at 1CF28. In the meantime, S/G 1A level continued to decrease. Before

Instrumentation and Electrical (IAE) personnel could repair the leaking manifold, the control air leakage caused 1CF28 to close at 0717 hours. CROs reduced Reactor power to 86% in an attempt to match CF flow with demand.

CROs manually tripped the Unit, from 86% Reactor power, at 0718:03:201 hours, when it was realized that a S/G 1A Low Low Level Reactor Trip signal would occur. Emergency Procedure EP/1/A/5000/01, Reactor Trip or Safety Injection, was entered. At 0718:03:261 hours, the Turbine tripped as a result of the Reactor trip. The S/G 1A Low Low Level Reactor Trip signal was received at 0718:04:067 hours. Both Motor Driven CA pumps automatically started on low low S/G 1A level. All appropriate S/G Blowdown EIIS:WI! (BB) and Nuclear

Sampling EIIS:KN! (NM) valves closed as expected. A Feedwater Isolation occurred due to a Reactor trip concurrent with low Tave.

The post trip cooldown was controlled by the automatic cycling of the Bank 2 and Bank 3 Main Steam Bypass to Condenser EIIS:SO! (SB) valves. In addition, S/G Power Operated Relief Valves (PORVs) 1SV19, S/G A PORV, 1SV13, S/G B PORV, and 1SV1, S/G D PORV, actuated when Main Steam (EIIS:SB! (SM) pressures increased due to the Bank 1 SB valves not opening. 1SV7, S/G C PORV, also opened, but was blocked at the time due to modification work in progress. Work Request 50915 OPS was initiated to investigate and repair the Bank 1 SB valves. The CROs took manual control of CA and stabilized S/G levels. Unit 1 was stabilized in Mode 3, Hot Standby. The CF, CA, NM, and BB Systems were subsequently realigned and Unit 1 entered Mode 1 on June 27 at 0340 hours.

Work Request 50912 OPS allowed for the replacement of 1CF28's control air manifold gasket by IAE. The gasket was found to be torn between two port holes located in the gasket. 1CF28 had been last worked on June 17, 1989, on Work Request 01232 MES. It was found, at that time, that the control air manifold gasket was brittle apparently due to normal wear, and was replaced. The new gasket was apparently torn during installation, which was the cause of this incident. The three other Unit 1 S/G Main Feedwater Control valves were inspected post trip. 1CF37, S/G 1B CF Control Valve, and 1CF46, S/G 1C CF Control Valve, were found to have small leaks. These valves were repaired per Work Requests 1259 MES and 1260 MES, respectively.

CONCLUSION

This incident has been attributed to inappropriate actions, due to actions being performed with insufficient precision. The gasket installed under Work Request 01232 MES was apparently damaged prior to or during installation.

Train A CA Flow did not reset at its normal setpoint (S/G level of 17%). CA did not reset until S/G 1A level had increased to approximately 40%. Work Request 50914 OPS was issued and the reset circuit time delay has been recalibrated.

The investigation of the Bank 1 SB valves revealed a blown fuse. The fuse has been replaced per Work Request 50915 OPS.

Another deficient aspect of the post transient response was the failure of the Digital Rod Position Indication (DRPI) EIIS:AA! (EDA) System to indicate that the control rod EIIS:ROD! D-8, in Shutdown Bank E, was fully inserted following the Reactor trip. The DRPI System indicated the rod to be at six steps withdrawn. Work Request 50913 OPS replaced a defective encoder card, which corrected the indication problem.

There was a question, during the post trip evaluation, as to whether 1SV13 responded properly. The alarm printer indicates the valve went open for approximately 16 seconds and the Transient Monitor indicated that it actuated at a pressure of 1127 psig. In order to verify the operability of 1SV13, PT/1/A/4200/31, S/G PORV Stroke Test, was satisfactorily performed. In addition, modifications described in Work Request 3287 NSM, to improve the performance of 1SV13, were completed after the trip.

A review of incidents in the past twelve months indicates there have been no previous Engineered Safety Feature actuations due to actions performed with insufficient precision. Therefore this is not considered to be a recurring event.

The manifold assembly is manufactured by Moore Products Company. The positioner for 1CF28 is model number 7214315 and the gasket is part number 118012.

The blown fuse causing the Bank 1 SB valves to not actuate was supplied by Westinghouse, and is identified as part number 743A407H06.

CORRECTIVE ACTION

IMMEDIATE

(1) CROs placed 1CF28 and 1CF30 in manual and fully open, to provide additional CF flow.

SUBSEQUENT

(1) An Operator was sent to inspect 1CF28.

(2) CROs reduced Reactor power and increased CFPT 1A speed.

(3) CROs manually tripped the Reactor and subsequently stabilized Unit 1 in Mode 3, Hot Standby, in accordance with EP/1/A/5000/01.

(4) IAE replaced the damaged gasket under Work Request 50912 OPS.

(5) IAE inspected the other Unit 1 S/G CF Control Valve positioners and repaired 1CF37 and 1CF46 leaks under Work Requests 1259 MES and 1260 MES, respectively.

(6) 1SV13 was successfully tested under PT/1/A/4200/31A, S/G PORV stroke Test, and modifications per 3287NSM were completed.

(7) IAE replaced a defective encoder card on DRPI under Work Request 50913 ops.

(8) IAE recalibrated the reset circuit time delay for Train A CA under Work Request 50914 OPS.

(9) IAE replaced a blown fuse under Work Request 50915 OPS, to repair the Bank 1 SB valves response.

PLANNED

(1) The inspection of the CF control valves will be incorporated into the preventive maintenance program.

(2) The responsibility for resolution of all minor, inadequate post-trip

responses not required to be resolved prior to Unit startup will be assigned and all such items will be identified on the Station Commitment Index.

SAFETY ANALYSIS

Upon closure of 1CF28, the Operator initiated a manual Unit runback by reducing Turbine/generator (EIIS:GEN! load with the control rods in automatic.

Over approximately 1 minute and 24 seconds, Reactor power was reduced to 86%. A manual Reactor trip was initiated at 86% full power in anticipation of a S/G 1A Low Low Level Reactor trip. The Reactor trip breakers EIIS:BRK! opened within 52 milliseconds of the manual Reactor trip signal. All of the control rods descended to the bottom of the core, reducing power to decay heat level. Approximately 1 second after manual Reactor trip, S/G A Low Low Level signal occurred, autostarting the Motor Driven CA pumps. A Feedwater isolation occurred upon Reactor trip with low Tave (564 degrees F).

Reactor Coolant System Loop A temperature increased 2 degrees F upon closure of 1CF28 to a value of 592 degrees F, and then decreased during the Unit runback to 589 degrees F at the time of Reactor trip. Reactor Coolant System Loops B, C, and D temperature decreased during the Unit runback to approximately 587 degrees F at the time of Reactor trip. Reactor Coolant System temperature then decreased to a minimum value of 553 degrees F post-trip, and stabilized at 558 degrees F 30 minutes post-trip, 1 degree F from the no-load target of 557 degrees F. Pressurizer pressure decreased during the Unit runback from 2235 psig to 2222 psig at the time of Reactor trip. Pressurizer pressure then decreased to a minimum value of 2028 psig post-trip, and stabilized at the no-load target of 2235 psig within 30 minutes

post-trip. Pressurizer level decreased from 61% to 58% during the Unit runback prior to the trip.

Pressurizer level decreased to a minimum level of 20% post-trip, and stabilized at 30% within 30 minutes post-trip, 5% from the no-load target of 25%. Steam pressure in S/Gs 1A, 1B, 1C, and ID reached maximum values of 1121 psig, 1147 psig, 1131 psig, and 1128 psig, respectively. Average steam pressure stabilized at 1080 psig within 30 minutes post-trip, 10 psi from the no-load target of 1090 psig. S/Gs 1B, 1C, and 1D narrow range levels

remained on-scale post-trip. S/G 1A wide range level decreased to a minimum indicated value of 35% post-trip. Correction of this value for calibration condition variations yields an actual minimum level of 45%. S/G narrow range levels stabilized at 35% within 30 minutes post-trip.

Prior to the trip, Bank 1 SB valves opened for approximately 30 seconds to dump steam to the condenser and reduce Reactor Coolant System Tave (the valves

were controlling in the Tave mode). Banks 2 and 3 SB valves opened post-trip to dump steam to the condenser. Bank 1 SB valves malfunctioned and remained closed post-trip. This lack of availability of Bank 1 SB valves contributed to the steam pressure increase and caused 1SV19, 1SV7, and 1SV1, the PORVs for S/Gs A, C, and D, respectively, to open as designed upon pressure increase

above 1125 psig. 1SV19 and 1SV1 relieved steam to the atmosphere, and the S/G C block valve for the PORV was closed for modification work to be performed on the PORV. Therefore, 1SV7 did not relieve steam. Based on computer alarm typer information and steam pressure plots, 1SV13, S/G 1B PORV, apparently opened slightly and then closed within several seconds, not enough to impact steam pressure.

As required by the Reactor Trip Emergency Procedure, CA flow was maintained greater than 450 gpm while S/G wide range level indication was less than 47%. The Reactor Coolant was 68 degrees F subcooled at the point of minimum Pressurizer pressure. Adequate core decay heat removal was available and maintained at all times. Because of a timer setpoint drift, the CA System reset was not successfully accomplished on the initial attempt and CA flow to the S/Gs was not throttled until approximately 15 minutes after initiation of the transient and at a S/G narrow range level of 47%. However, no overcooling concerns existed and Unit no-load parameter targets were approached. Additionally, credit is not taken for this interlock in any accident scenario. For example, in the MSLB scenario, the CA System is assumed to operate at full flow assuming a single active failure.

The Unit is designed to accommodate a conservative set of initial conditions (for Condition II, III, and IV events) corresponding to adverse conditions which can occur during Condition I operation (normal operational transients).

One such normal operational transient is a ramp load change of 5% per minute.

In this event, the Unit ranback 13% in 84 seconds. However, the runback was not a separate operational transient, but part of the event which is bounded by the "Loss of Normal Feedwater Flow" transient as discussed in Section 15.2.7 of the

FSAR. This event was less severe than a total loss of feedwater as assumed in the FSAR. Also, the FSAR assumes operability of only the S/G code safety valves. In this event, S/G PORVs and steam dump to condenser valves were available and used to remove heat.

All safety related systems and equipment were available throughout this event.

The cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, Reactor Coolant System, and containment structure was maintained at all times. The health and safety of the public were not affected by this event.

ATTACHMENT 1 TO 8908010458 PAGE 1 OF 1

Duke Power Company (803) 931-3000
Catawba Nuclear Station
P.O. Box 256
Cloter, S.C. 29710

DUKE POWER

July 26, 1989

Document Control Desk
U. S. Nuclear Regulatory commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
LER 413/89-17

Gentlemen:

Attached is Licensee Event Report 413/89-17, submitted concerning manual reactor trip due to torn gasket of main feedwater valve positioner control air manifold.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Tony B. Owens
Station Manager

KEB\LER-NRC.TBO

xc:
Mr. S. D. Ebneter American Nuclear Insurers
Regional Administrator, Region II c/o Dottie Sherman, ANI Library
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101 Marietta Street, NW, Suite 2900 270 Farmington Avenue
Atlanta, GA 30323 Farmington, CT 06032

M & M Nuclear Consultants Mr. K. Jabbour
1221 Avenues of the Americas U. S. Nuclear Regulatory commission
New York, NY 10020 Office of Nuclear Reactor Regulation
Washington, D. C. 20555

INPO Records Center
Suite 1500 Mr. W. T. Orders
1100 Circle 75 Parkway NRC Resident Inspector
Atlanta, GA 30339 Catawba Nuclear Station

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